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Appendix IV: 10-Yr Plan for the Supercritical Water Reactor

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SUPERCRITICAL LIGHT WATER-COOLED REACTOR (SCWR)

Supercritical water-cooled reactors (SCWRs) are promising advanced nuclear systems because of their high thermal efficiency (i.e., about 45% vs. about 33% efficiency for current Light Water Reactors, LWRs) and considerable plant simplification. SCWRs are basically LWRs operating at higher pressure and temperatures with a direct once-through cycle. Operation above the critical pressure eliminates coolant boiling, so the coolant remains single-phase throughout the system. Thus the need for recirculation and jet pumps, pressurizer, steam generators, steam separators and dryers is eliminated. The main mission of the SCWR is generation of low-cost electricity. It is built upon two proven technologies, LWRs, which are the most commonly deployed power generating reactors in the world, and supercritical fossil-fired boilers, a large number of which is also in use around the world. The SCWR concept is being investigated by 32 organizations in 13 countries. General information about the SCWR concept and its technical challenges is widely available in the literature, and will not be repeated here.

The Generation-IV International Forum (GIF) SCWR Steering Committee has generated a schedule for the demonstration of the SCWR concept that call for the completion of all essential R&D by 2015 and construction of a small-size (≤ 150 MWt) prototype SCWR by 2020.

The objective of this 10-year plan is to assess the technical feasibility of the SCWR concept. Therefore, the plan focuses on the two key feasibility issues that were identified in the Generation IV Roadmap Report for this concept, i.e., selection/development of structural materials, and demonstration of adequate safety and stability. Issues like economic evaluation, detailed design and materials codification are deemed of secondary importance at this stage, and thus are not addressed.

1 THE U.S. REFERENCE SCWR PROGRAM

In the U.S. the Generation-IV SCWR program operates under the following general assumptions, which are consistent with the SCWR's goal of electricity generation at low capital and operating costs:

- ◆ Direct cycle,
- ◆ Thermal spectrum,
- ◆ Light-water coolant and moderator,
- ◆ Low-enriched uranium oxide fuel,
- ◆ Base load operation.

The reference SCWR design for the U.S. program is a direct cycle system operating at 25.0 MPa with core inlet and outlet temperatures of 280 and 500°C, respectively. The coolant density decreases from about 760 kg/m³ at the core inlet to about 90 kg/m³ at the core outlet. The inlet flow splits with about 10% of the inlet flow going down the space between the core barrel and the reactor pressure vessel (the downcomer) and about 90% of the inlet flow going to the plenum at the top of the reactor pressure vessel to then flow downward through the core in special water rods to the inlet plenum. Here it mixes with the feedwater from the downcomer and flows upward to remove the heat in the fuel channels. This strategy is employed to provide good moderation at the top of the core. The coolant is heated to about 500°C and delivered to the turbine. The reference power, efficiency, pressure, and coolant flow rate and temperatures are listed in Table I. Figure 1 is a sketch of the reactor pressure vessel and internals showing the coolant flow paths. The components limiting the power rating of the SCWR are the turbine and the reactor pressure vessel.

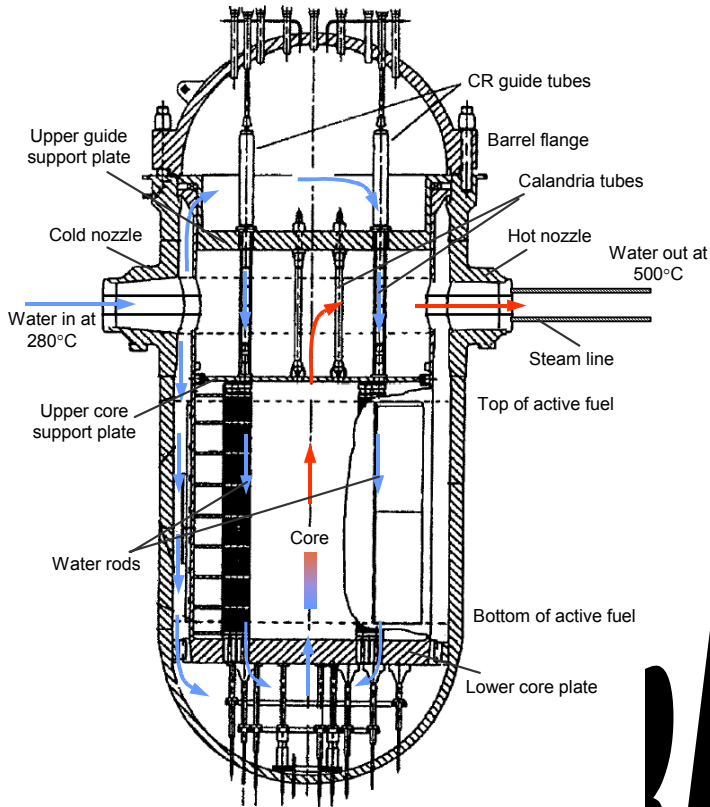


Table I. U.S. Generation-IV SCWR reference design power and coolant conditions.

Parameter	Value
Thermal power	3575 MWt
Net electric power	1600 MWe
Net thermal efficiency	44.8%
Operating pressure	25 MPa
Reactor inlet temperature	280°C
Reactor outlet temperature	500°C
Reactor flow rate	1843 kg/s
Plant lifetime	60 years

Figure 1. The SCWR reactor pressure vessel.

1.1 SCWR Pressure Vessel

Key dimensions for the current SCWR vessel are listed in Table II. The vessel will be exposed to 280°C inlet coolant on the inside surfaces. The outlet nozzles will be protected with a 2" thermal sleeve, which maintains the nozzles below 350°C. Peak fluence of the RPV is expected to be no more than $5 \times 10^{19} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$).

Table II. Reference reactor pressure vessel design for the U.S. Generation-IV SCWR

Parameter	Value
Type	PWR with top CRDs
Height	12.40 m
Material	SA-508
Operating/design press.	25.0/27.5 MPa
Operating/design temp.	280/371°C
# of cold/hot nozzles	2/2
Inside diameter of shell	5.322 m
Thickness of shell	0.46 m
Inside diameter of head	5.352 m
Thickness of head	0.305 m
Vessel weight	780 t
Peak fluence ($>1 \text{ MeV}$)	$<5 \times 10^{19} \text{ n/cm}^2$

1.2 SCWR Core and Fuel Assembly Design

The reference SCWR core design is shown in Figure 2. The core will have 145 assemblies with an equivalent diameter of about 3.9 meters. The core barrel will have inside and outside diameters of about 4.3 and 4.4 meters, respectively.

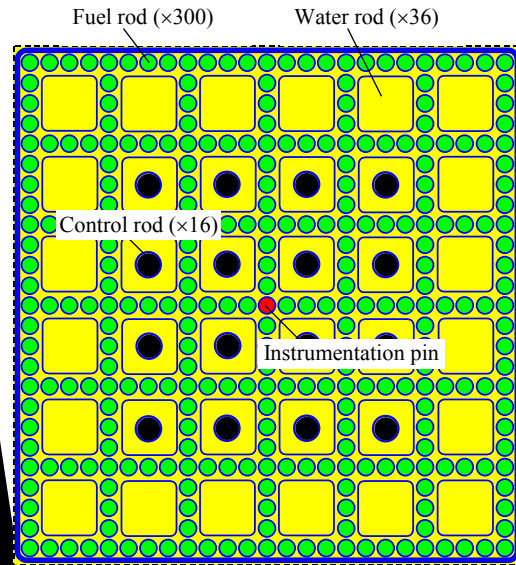
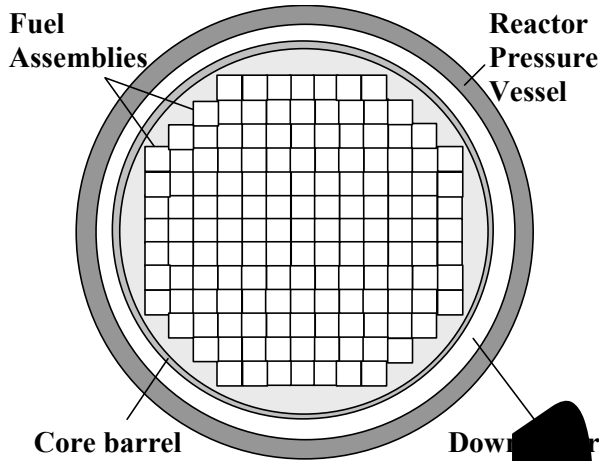


Figure 2. Sketch of the reference SCWR core. Figure 3. The SCWR fuel assembly with water rod boxes.

The reference SCWR fuel assembly design is shown in Figure 3 and the relevant dimensions are listed in Table III. It may be necessary to insulate the water moderator boxes to retain a sufficient moderator density, as well as portions of the vessel internals supplying water to the core.

Table III. Reference fuel assembly design for the U.S. Generation-IV SCWR

Parameter	Value
Fuel pin lattice	Square 25x25 array
Number of fuel pins per assembly	300
Number of water rods per assembly	36
Water rod side	33.6 mm
Water rod wall thickness	0.4 mm (plus insulation if needed)
Number of instrumentation rods per assembly	1
Number of control rod fingers per assembly	16
Active control rod materials	B ₄ C for scram, Ag-In-Cd for control
Number of spacer grids	14 (preliminary estimate)
Assembly wall thickness	3 mm (plus insulation if needed)
Assembly side	286 mm
Inter-assembly gap	2 mm
Assembly pitch	288 mm

Average power density	69.4 kW/L
Average linear power	19.2 kW/m
Peak linear power at steady-state conditions	39 kW/m

The reference fuel pin dimensions are listed in Table IV. With the exception of the plenum length and fill pressure, the fuel pin dimensions are typical of 17 by 17 PWR fuel assembly pins. However, the fuel pin pitch is considerably smaller than the pitch used in LWRs. The U-235 enrichment, the Gd_2O_3 loading and fuel burnup are typical of the values used in high burnup LWR fuel.

Table IV. Reference fuel pin dimensions for the U.S. Generation-IV SCWR

Parameter	Value
Fuel pin outside diameter	10.2 mm
Fuel pin pitch	11.2 mm
Cladding thickness	0.63 mm
Heated length	4.27 m
Fission gas plenum length	0.6 m
Total fuel pin height	4.87 m
Fill gas pressure at room temperature	6.9 MPa

1.3 SCWR Pressure Vessel Internals

The important RPV internals include the lower core support plate, the core former, the core barrel, the upper core support plate, the calandria tubes located immediately above the upper core support plate, the upper guide support plate, the hot nozzle sleeve or insulation, and the control rod guide tubes. The location and approximate shape of each of these components is shown in Figure 1.

Some of these components, including the lower core support plate and the control rod guide tubes in the upper head, will be subjected to normal PWR coolant temperature conditions and will be similar to the components typically used in PWRs. However, a number of the RPV internals, including the core barrel (or possibly the core former), the upper guide support plate, the calandria tubes, and the RPV hot nozzle sleeve, will be in contact with water at the inlet temperature at 280°C on one side and water at the hot outlet coolant at a temperature of 500°C on the other side.

1.4 SCWR Coolant System

The SCWR reactor coolant system has two feedwater lines and two steam lines (as opposed to four in a BWR of similar thermal power). The main parameters of the SCWR reactor coolant system are reported in Table V.

Table V. SCWR reactor coolant system parameters.

Parameter		Value
Feedwater lines	Number	2
	Operating temperature	280°C
	Operating/design pressure	25/27.5 MPa
	OD/thickness	400 mm / 51 mm
Steam lines	Number	2
	Operating temperature	500°C

	Operating/design pressure	25/27.5 MPa
	OD/thickness	470 mm / 51 mm

1.5 SCWR Power Conversion System

The reference SCWR system will have a power conversion cycle that is very similar to a supercritical coal-fired plant, with the boiler replaced by the nuclear reactor. The cycle is based on a large single-shaft turbine with one high-pressure/intermediate-pressure unit and three low-pressure units operating at reduced speed (1800 rpm). The steam parameters at the high-pressure/ intermediate-pressure unit inlet are 494°C and 23.4 MPa, well within current capabilities of fossil plants. Similarly to traditional light water reactor (LWR) cycles, a moisture separator-reheater (MSR) module is located between the high-pressure/intermediate-pressure and the low-pressure turbines, and reheating is achieved with live nuclear steam. Heat rejection occurs in traditional natural-draft cooling towers. Eight feedwater heaters raise the condensate temperature to the reactor inlet level of 280°C. The main feedwater pumps are turbine-driven and operate at about 190°C.

2 SYSTEM DESIGN & EVALUATION

It is envisioned that the first phase of the Gen-IV R&D program for SCWR will focus on demonstrating the technical feasibility of the concept. This phase will address five critical issues as identified in the Generation-IV Roadmap report:

- 1) Establish a baseline design for the SCWR core and reactor coolant system,
- 2) Generate basic data on heat transfer, pressure drop and critical flow for supercritical water at SCWR prototypical conditions,
- 3) Identify suitable safety systems and containment designs to cope with the consequences of major abnormal events,
- 4) Evaluate the susceptibility of the SCWR to thermal-hydraulic and coupled thermal-hydraulic/neutronic instabilities,
- 5) Develop a strategy for reactor control including start-up and operational transients.

Although task 2 logically precedes all others, information on the thermal-hydraulics of supercritical water in fossil-fueled power plants already exists that will enable starting tasks 1, 3, 4 and 5 without waiting for the generation of an ad-hoc database.

2.1 Baseline Design for the SCWR Core and Reactor Coolant System

The objective of this task is to establish a baseline conceptual design for the core and reactor coolant system to which all other activities (e.g., materials, safety, etc.) can relate. Activities will include development of a credible thermal, neutronic and mechanical design for the fuel assembly and vessel internals to provide adequate moderation in the core, development of a reactivity control system based on control rods and burnable poisons, and design of a suitable power conversion cycle. The operating temperatures and pressure, linear heat generation rates, core geometry, flow rates and power-conversion cycle conditions will be identified. Several projects already exist that include significant design activities for the SCWR. The University of Tokyo with the help of vendors Toshiba and Hitachi has performed a complete conceptual design of a large SCWR plant. The European Commission has sponsored a three-year project on SCWRs involving various research laboratories and the vendor Framatome-ANP, which has also resulted in a reference SCWR design. Canadian vendor AECL has developed a supercritical

version of their CANDU system, while in Korea KAERI has initiated a feasibility study to upgrade their standard KNP design to supercritical conditions. In the U.S. there are two NERI projects launched in 2001 that include significant design activities: one is led by the INEEL and involves the Westinghouse Electric Company, the other is led by the University of Wisconsin at Madison and involves the Argonne National Laboratory (ANL). Both projects have produced significant modifications to the Japanese and European baseline designs.

A comparative analysis of these various core designs will be made to evaluate the relative merits and shortcomings of each, and their potential to meet the Generation-IV goals; the objective is to converge on a design that can be jointly developed and eventually demonstrated in cooperation with other GIF countries.

2.2 Basic Thermal Data for the SCWR

Because of the lack of phase change in the core SCWRs, unlike LWRs, cannot use design criteria based on the critical heat flux concept. The commonly accepted practice is to specify cladding temperature limits that must be met during different events. This makes it very important to predict the heat transfer coefficient to the supercritical water coolant with great accuracy. However, while considerable information exists on heat transfer to supercritical water in round tubes for fossil boilers, little is known about the effect of the geometry and flow conditions typical of the SCWR core. Therefore, this project addresses the critical issue of predicting heat transfer to supercritical water at prototypical SCWR conditions and to develop the tools to predict the SCWR thermal behavior.

Both actual SCW and supercritical surrogate fluids (CO₂ and freon) will be explored in this task. Surrogate fluids are convenient because some existing facilities already use such fluids, which in general have considerably lower critical pressures and temperatures, thus affording significant cost and time savings in constructing and operating the experimental facilities. On the other hand SCW provides a direct representation of the SCWR behavior without the need for scaling of the thermo-physical properties.

The experiments should cover the various heat transfer regimes expected during operation of the SCWR, including upflow, downflow and horizontal forced convection at high and low mass fluxes, buoyancy assisted forced convection, pure free convection, and deteriorated heat transfer. Transition from one regime to another including the depressurization to two-phase flow conditions of an initially supercritical fluid should be correlated in terms of dimensionless groups facilitating the comparison among different fluids, geometries and flow conditions. Also, in bundle geometry where a circumferential symmetry does not exist, provisions should be made to measure the azimuthal variation of the heat transfer coefficient. Special effects such as flow channel shape, grid spacers, and non-uniform heat flux should be quantified.

The surrogate fluid work will consist of the following elements:

- ◆ Current facilities are suitable for single-tube or single-rod experiments only. It will be necessary to upgrade existing CO₂ and freon facilities or construct a new CO₂ facility to accommodate a rod-bundle test section.
- ◆ Measure the heat transfer coefficient in single-tube and single-rod experiments to establish a connection with the fossil boiler database.
- ◆ Measure the heat transfer coefficient in bundle experiments.

To simulate the SCWR core, the SCW heat transfer facility will have the following requirements: pressures up to 27 MPa, bulk water temperatures up to 550°C, surface heat fluxes of up to 1.5 MW/m²

with various axial power shapes, and a test section with a bundle of heated rods of proper and variable geometry. There exists a SCW facility at the Framatome-ANP laboratories in Erlangen, Germany (see Figure 4) that was used for supercritical fossil boiler tube tests and could be used by GIF for single-tube experiments now. This facility also has a sufficiently large power supply and pump to accommodate a relatively large heated-rod bundle. However, the actual bundle test section would have to be built as part of this project. The SCW work will then consist of the following elements:

- ◆ Upgrade of the Erlangen facility (the U.S. will design and construct the bundle test section).
- ◆ Measure the heat transfer coefficient at prototypical SCWR flow and geometry conditions.

In parallel with the experimental work, interpretation of the experimental data including scaling effects for different fluids, geometries and flow conditions will be performed. This work includes the development and validation of best-estimate heat transfer correlations and models to predict the heat transfer coefficient in the SCWR core.

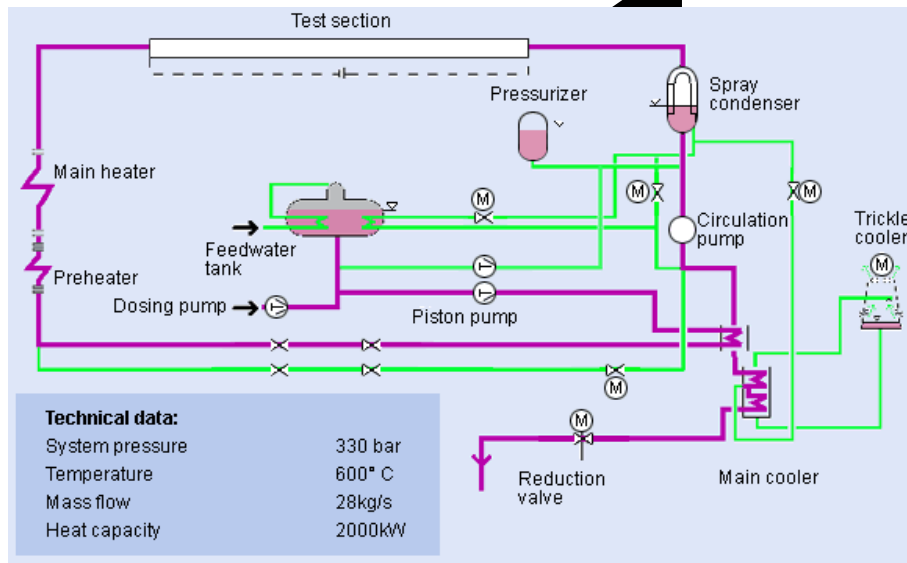


Figure 4. Flow diagram of the Benson test rig in Erlangen.

There is also a lack of data for critical (or choked) flow at supercritical conditions since LWRs (for which most critical flow work has been performed in the past) operate at subcritical pressures. Critical (or choked) flow phenomena are of great importance in designing/operating the reactor safety/relief valves and the automatic depressurization system, as well as in the analysis of LOCA events.

A facility will be constructed consisting of a pressure tank, a discharge nozzle, various valves, measuring equipment, and data acquisition equipment. The design pressure and temperature will be <30 MPa and 500°C. The stagnation conditions in the tank as well as the diameter and length of the discharge nozzle will be systematically varied. Direct experimental measurements of the temperature and pressure along the discharge nozzles, and of the void fraction and flow rate at the nozzle outlet will be obtained. These data will enable accurate benchmark of existing critical-flow models and, if needed, development of new ones. The schematic diagram of a possible design for this test facility proposed by the University of Wisconsin at Madison is shown in Figure 5. The pressure vessel will be mounted to the ceiling and allowed to pivot on bearing assemblies to allow free movement opposite to the mass ejection. The momentum of the discharge will be measured by a force transducer on the side of the facility, since this

will be free to move sideward.

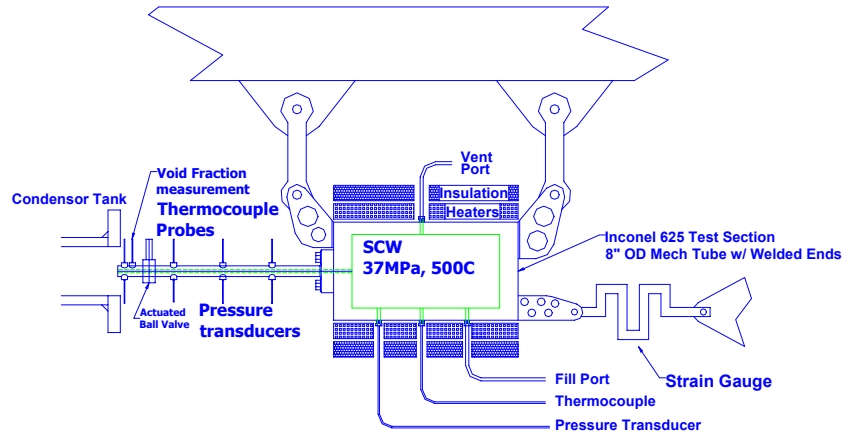


Figure 5. Supercritical water flow loop facility.

These basic data on heat transfer and critical flow will have to be rationalized and incorporated into existing computational codes such as RELAP5, FLUENT, etc., for use in the SCWR analysis. The R&D needed to accomplish this is described in the next section and appendix of this report (See Hussein's sections).

2.3 SCWR Safety Systems and Containment Design

Because the SCWR eliminates the need for an indirect cycle with steam generators as well as for coolant recirculation within the vessel, the thermal inertia of the primary system is significantly lower than for an LWR. Therefore the question of how the system responds to accident events such as LOCAs or LOFAs becomes important.

A conceptual design and analysis of the SCWR containment and safety systems will be performed. This will include the reactor protection system, the residual heat removal system, the overpressurization protection system, the ECCS, reactor shutdown system, steam and pressure relief systems, etc. A general safety strategy to cope with postulated sequences (e.g., depressurization vs. high-pressure emergency coolant injection) will be defined. It will be necessary to determine if this strategy can be implemented with active or passive safety systems. This task will mostly focus on assessing the applicability to the SCWR of passive safety systems developed for advanced LWRs (e.g., ESBWR, AP-600) including isolation condensers, gravity-driven cooling systems and passive containment cooling system.

2.4 SCWR Stability Analysis

SCWRs present the possibility of various types of instabilities, namely, density-wave instabilities, coupled thermal-hydraulic/neutronic instabilities, and natural circulation instabilities. It is necessary for any given design to show that either the oscillations do not occur during normal operation or that if they

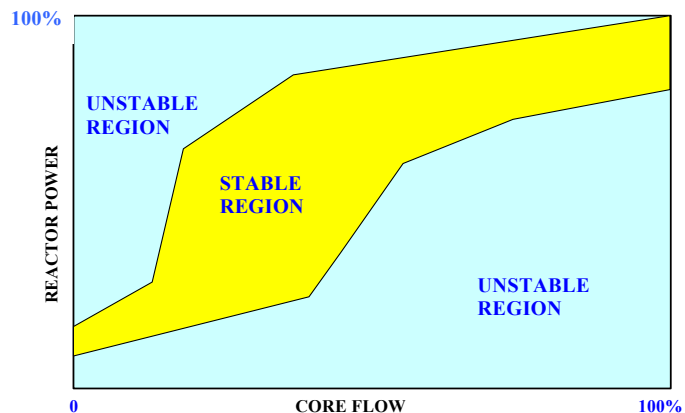


Figure 6. Conceptual Envelope for Stable Power-Flow Operation of SCWR

do, they can be detected and suppressed in a safe manner. Finally, oscillations under accident conditions must also be considered, e.g., under anticipated transient without scram conditions. The objective of this task is a better understanding of instability phenomena in SCWRs, the identification of the important variables affecting these phenomena, and ultimately the generation of maps (a conceptual example is shown in Figure 6) identifying the stable operating conditions of the different SCWRs designs. Consistent with the U.S. NRC approach to BWRs licensing, the licensing of SCWRs will probably require, at a minimum, demonstration of the ability to predict the onset of instabilities. This can be done by means of a frequency-domain linear analysis. Prediction of the actual magnitude of the unstable oscillations beyond onset, although scientifically interesting and relevant to beyond-design-basis accidents, will likely not be required for licensing and can be delayed to a second phase of the SCWR development.

Therefore, in this task simplified analytical models will be developed to predict the onset of instability of the density-wave, coupled thermal-hydraulic/neutronic and natural-circulation type. The models will capture the effect of important variables such as axial and radial power profile, moderator density and fuel temperature reactivity feedback, fuel rod thermal characteristics, coolant channel hydraulic characteristics, heat transfer phenomena, core boundary conditions (including the effect of direct or indirect cycles), etc. Mitigating effects like emergency insertion of control rods, and fuel modifications to obtain appropriate thermal and/or neutronic response time constants will also be assessed using analytical simulations. Parallel channel instabilities will be investigated as well as instabilities during start-up and partial load operation. Also, existing supercritical water and/or CO₂ loops will be used to perform experiments on both natural circulation as well as density-wave type instabilities. These facilities will provide valuable data for benchmarking the analytical models.

2.5 SCWR Control and Start-up

An important issue is that of control of the main reactor variables, e.g., core power, coolant pressure and temperature. The University of Tokyo group designed a control system for their direct-cycle fast reactor in which the core power is controlled by the control rods, the pressure by the turbine throttle valve, and the coolant core outlet temperature by the pump flow. However, other approaches are possible and could be better. Equally important is the issue of plant start-up from cold conditions. In fossil-fueled supercritical plants two different start-up approaches are possible including one with constant pressure and one with sliding pressure. In a SCWR implementation of these approaches will likely require installation of dedicated out-of-vessel components (e.g., a flash tank for constant-pressure start-up, a steam separator for sliding-pressure start-up). Therefore, besides the technical feasibility of these start-up approaches, this task will have to assess the cost impact of one versus the other, and the sizing of these components. The conceptual design of the reactor control and start-up systems will be performed by an industrial organization with experience in the fossil power industry.

3 MATERIALS ISSUES AND REQUIREMENTS

Only the materials requirements are reported here. The actual R&D needed to select and/or develop materials that meet these requirements is described elsewhere in this report (see Section..., and Appendix..., i.e., Corwin's sections). However, this appendix does report the budget for all SCWR materials R&D.

3.1 Reactor Pressure Vessel

The inner surface of the vessel will be exposed to water at 280°C thus would be clad with a weld overlay of Type 308 stainless steel and the outer surface will be insulated, most likely in a manner similar to existing PWRs. Given the operating temperature of 280°C and an expected irradiation exposure similar to that of current generation pressurized water reactors (PWR), the primary candidate materials for the RPV shell are those currently used in PWRs, namely SA 508 Grade 3 Class 1 forging (formerly designated SA 508 Class 3) or SA 533 Grade B Class 1 plate. The RPV thickness given above assumes one of those materials. Of those two materials, which have similar chemical compositions and the same design stress intensities in the ASME Code, the SA 508 Grade 3 Class 1 forging is preferred to eliminate the need for axial welds. It is also desirable to fabricate a forging of sufficient height to keep circumferential welds outside the region adjacent to the reactor core (the so-called beltline region) and preliminary information from the Japan Steel Works indicates that it will probably be possible to do so.

The knowledge gained over the past few decades regarding the radiation embrittlement of current LWR materials must be utilized in the preparation of the material specifications for the RPV materials. For example, minimization of sensitizing elements such as copper and phosphorus is critical for mitigation of embrittlement and undesirable segregation, while the nickel content should be kept relatively low yet high enough to maintain the strength and fracture toughness of the A508 Grade 3 Class 1 steel. In this regard, the thickness of the SCWR vessel shell and nozzle course forgings may present difficulties. Therefore, special attention must be paid to the chemical composition and heat treatment specifications to allow for through-thickness hardening to maintain the necessary strength and fracture toughness, yet to also ensure minimization of irradiation embrittlement sensitivity.

Similar to the RPV shell, the RPV bottom, nose head and welded bottom head will operate at 280°C and the materials of construction will be similar. The materials and fabrication of the heads, including the control rod drive mechanism housings, tie bolts, etc. will incorporate the latest materials of choice for current LWRs and currently designed advanced LWRs. Information regarding RPV supports is not yet available and the choice of materials will depend upon the specific design.

3.2 RPV Internals

Three factors will most affect the properties and choice of the structural materials from which the SCWR components will be fabricated. These are effects of irradiation, high-temperature exposure, and interactions with both the sub- and supercritical water environment to which they are exposed. An extensive testing and evaluation program will be required to assess the effects that these factors have on the properties of the potential materials for SCWR construction to enable a preliminary selection of the most promising materials to be made and to then qualify those selected for the service conditions required. Tables VI and VII identify the performance requirements (i.e., the anticipated irradiation conditions and mechanical loads for normal operating conditions, as well as the temperature excursions expected for abnormal conditions) and candidate materials for the fuel assembly components and other vessel internals, respectively. The first category includes the fuel cladding, fuel rod spacers (spacer grid or wire wrap), water rod boxes, fuel assembly ducts, and control rod guide thimbles. The second category includes control rod guide tubes, the upper guide support plate (UGS), calandria tubes, upper core support plate (UCS), lower core plate (LCP), core former, core barrel, and threaded structural fasteners. Also listed are materials typical of those in use for similar components in currently operating pressurized water reactors (PWRs) and boiling water reactors (BWRs).

Once a limited set of candidate alloys will be downselected for the cladding, a series of tests will be done to evaluate the safety limits of the fuel pin. Pressure burst and ballooning tests will simulate the fuel

pin behavior during depressurization following a large LOCA; rapid heat-up tests will be needed to simulate RIAs, etc.

3.3 Pump, Piping, and Valve

The issues and concerns regarding the pumps, valves, and piping for the SCWR can be divided into those associated with the feedwater lines and the steam lines.

Issues for components of the feedwater system will be similar to those being considered in the more conventional advanced LWR technologies, where ASME Section III is the applicable construction code. Experience has shown that flow-assisted corrosion (FAC) is the dominant degradation mechanism of LWR piping system. Also, fatigue and stress corrosion cracking are concerns. Carbon steels piping materials in operating LWRs, such as seamless pipe SA-106 Grade C, clad carbon steels, and seamless

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Table VI. Operating conditions and candidate materials for the in-core reactor components of the SCWR. All components listed are part of replaceable fuel assembly.

Component	Normal Conditions			Abnormal Conditions	Current LWR Mtls		Candidate SCWR Materials	Notes
	Temperature ¹	Peak Dose ²	Loads ³	Temperature ⁴	PWR	BWR		
Fuel cladding	280-620 °C	15 dpa	Pressure drop across cladding, grid-cladding and fuel-cladding interactions σ up to 100 MPa	Up to 840°C for <30 sec	Zircaloy 4	Zircaloy 2	Fe-Ms, Low-swell S.S.	
Spacer grids/wire wrap	280-620 °C	15 dpa	Hold the fuel pins together	Up to 840°C for <30 sec	Zircaloy 4, Inconel 718	Zircaloy 4, Inconel X750, 304 S.S.	Fe-Ms, Low-swell S.S.	
Water rod boxes	280-300 °C inner 280-500 °C outer	15 dpa	$\Delta P < 0.1$ MPa	Up to 700°C for <30 sec	N/A	Zircaloy 2	Fe-Ms, Low-swell S.S.	May need to insulate.
Fuel Assembly duct	280-500 °C inner 280-300 °C outer	15 dpa	$\Delta P < 0.1$ MPa	Up to 700°C for <30 sec	N/A	Zircaloy 4	Fe-Ms, Low-swell S.S.	May need to insulate.
Control Rod Guide Thimble	280-300 °C	15 dpa	Low hydraulic and thermal stresses	280 - 300°C	Zircaloy 4	N/A	Zircaloy 4, Zr-Nb alloy	Zr alloy selected for superior neutron economy.
1. Peak temperatures in PWRs are 320-370°C 2. Design estimates for typical high burnup LWR fuel 3. In addition, all reactor internals will be subject to seismic and pipe break loads. 4 Condition II events only (LOCAs, LOFAs, ATWSs are excluded)			Fe-Ms (Ferritic-Martensitic) steels, e.g., T91 (9Cr-1Mo-V), A-21 (9Cr-TiC mod), NF616 (9Cr), HCM12A (12Cr), 9Cr-2WVTa, MA-957. Existing low-swell stainless steels, e.g., D-9 (14.5Cr-14.5Ni, 2Mo, Ti stab), PNC ~D-9 mod w/P).					

Table VII. Operating conditions and candidate materials for the core structural support reactor components of the SCWR.

Component	Normal Conditions			Abnormal Conditions	Current LWR Mtls		Candidate SCWR Materials	Notes
	Temperature ¹	Peak Dose ²	Loads ³	Temperature	PWR	BWR		
Upper Guide Support (UGS) plate	280 °C upper 500 °C lower	0.021 dpa	Significant hydraulic and thermal loads	Lower side at up to 700°C for <30 sec	304L S.S	304L S.S.	Advanced S.S., Fe-Ms	Must insulate between the region above the core (500°C) and the upper plenum (280°C) to limit the thermal loads in the UGS.
Calandria Tubes	280 °C inner 500 °C outer (w/o insulation)	0.021 dpa	Significant hydraulic and thermal loads	280°C inner 700°C outer	N/A	N/A	Advanced S.S., Fe-Ms	Must insulate to limit the heat transfer from the coolant to the moderator and control the thermal loads in the calandria tubes.
Upper Core Support (UCS) plate	500 °C	0.021 dpa	Significant hydraulic. Moderate thermal.	Up to 700 °C for <30 sec	304 S.S.	304, 304L, 316 S.S.	Advanced S.S., Fe-Ms	The water rod box penetrations may cause some locally high thermal stresses.
CR guide tubes	280 °C	0.00001 dpa	Low hydraulic. Low thermal.	N/A	304 S.S.	304 S.S.	Advanced S.S., Fe-Ms, 304L, 316L	May want to use the same material as for the UGS, UCS, and calandria tubes
Lower core plate	280-300 °C	0.39 dpa	Significant hydraulic. Low thermal. Supports core.	N/A	304L S.S	304L S.S.	Advanced S.S, Fe-Ms, 304L, 316L	May want to use the same material as for the UGS, UCS, and calandria tubes
Core Former	~280-600 °C	67.1 dpa	Significant hydraulic. High thermal.	700°C	304 S.S.	N/A	Fe-Ms, Low-Swell S.S.	Must insulate either the core former or core barrel to control the thermal loads in the barrel.
Core barrel or shroud	280°C core region, 500 °C above core	3.9 dpa	Significant hydraulic. High thermal.	N/A	304L S.S	304L S.S.	Fe-Ms, Low-Swell S.S.	Must insulate the core barrel above the core region and insulate either the core barrel or core former in the core region.
Threaded fasteners	280-500 °C	< 4 dpa ⁴			316 S.S./CW	304, 600, 316, 316L	Advanced S.S., IN-718, 625, 690	The current design is an all welded core former and barrel.

1. Peak temperatures in PWRs are 320-370°C

2. Design estimates for 60y

3. All reactor internals will be subject to seismic and pipe break loads

4. ~ 50 dpa for baffle bolts and formers in PWRs

Fe-Ms (Ferritic-Martensitic) steels, e.g., T91 (9Cr-1Mo-V), A-21 (9Cr-TiC mod), NF616 (9Cr), HCM12A (12Cr), 9Cr-2WVTa, MA-957.

Existing low-swell stainless steels, e.g., D-9 (14.5Cr-14.5Ni, 2Mo, Ti stab), PNC ~D-9 mod w/P). Advanced stainless steels, e.g., HT-UPS (~PNC), AL-6XN (20Cr-24Ni-6Mo-0.2Cu-0.2N), etc.

stainless steels pipes such as SA-312 TP304H, TP304L, TP316L are the primary candidate materials for the feedwater lines. Of these many materials, the grades that have been included in the LWR environmental strain-fatigue and fatigue crack growth studies would be preferred. Although seam welded piping has been installed in LWRs, it should be avoided unless the piping has been subsequently reworked and renormalized. Wrought products should be preferred over cast products.

The SCWR feedwater pumps will be low flow/high head pumps located on the feedwater lines outside the containment and are expected to operate at approximately 190°C. These pumps will resemble in many ways the state-of-the-art pumps developed for supercritical fossil power. The materials candidates for pump casing are a forged low-alloy steel, such as SA-508 Class 2 or Class 3. An austenitic cladding with controlled delta ferrite content would be required if a low-alloy steel is selected. Alternatively, an austenitic stainless steel such as SA-336 Gr F304 could be considered. The materials candidates for pump internals are a high-strength casting such as SA-487 CA-6NM-A (normalized and tempered 13Cr-4Ni steel).

The steam line piping is the greater concern. The issues related to the steam line system are more akin to those addressed in the design, construction and operation of supercritical fossil power plants. Creep and time-dependent material degradation are active in fossil plant steam line systems at temperatures above 370°C for ferritic steels and above 425 °C for austenitic alloys. The philosophy behind the ASME Power Piping Code (B31.1), which covers fossil plant piping, is significantly different from the philosophy of ASME Section III.

The outlet temperature of 500 °C is much higher than the temperature at which many supercritical fossil power plants operate, but the pressure (15.5 MPa) is comparable. Whereas ASME Section III has incorporated a wide selection of ferritic alloy steels for service to 370 °C and austenitic alloys for service to 425 °C the highest temperature extension Subsection NH is limited to Grade 22 Class 1, Grade 91, and three austenitic alloys (304H stainless steel, 316H stainless steel, and Alloy 800H). The steam line temperature is sufficiently low to enable the use of one of these materials, providing that FAC is not a problem. Alternate materials would include 316FR stainless steel. This steel qualifies as an "L" grade, yet has properties equivalent to, or superior to, Type 316H stainless steel. The database is sufficient to meet the needs for inclusion into Subsection NH.

The steam line piping system between the isolation valve and the turbine could be designed to meet the requirements of B31.1, which would allow a greater choice of materials, allowing the use of alloy P92 (9Cr-2W), which is used in fossil-fired supercritical plants. However, supplementary requirements to address fatigue and other damage accumulation mechanisms would be needed.

3.4 Power Conversion Cycle

The major components of the power conversion cycle are external to the reactor vessel and include the steam turbine and associated valving; the condenser; the demineralizer/condensate polisher; the feedwater preheaters; and the deaerator. There do not appear to be any special needs for alloy selection for the condenser, the demineralizer/condensate polisher, the feedwater preheaters, and the deaerator in the SCWR design, as long as the water chemistry guidelines developed for the control of corrosion in supercritical fossil plants can be followed. On the other hand, the turbine requires some special consideration.

Fossil-fired supercritical steam power plants operate with steam conditions typically of 540 to 600°C and 25 to 30.5 MPa. As a result, there is a well-established manufacturing base for turbines for operation at the supercritical steam conditions of interest in the SCWR, as well as extensive experience in their use. The extent to which this experience is relevant to the SCWR case largely depends on similarities and differences in the quality of the steam, in particular, the extent to which the level and types of impurities in the steam are different from those in fossil-fired practice. One difference is that, whereas in fossil-fired plants the steam exiting the high-pressure turbine is returned to the steam generator for reheating in a separate circuit before being sent to the intermediate-pressure and low-pressure turbines; reheating in LWRs is accomplished with live steam in order to minimize the complication of the steam circuit. Another modification adopted is the addition of moisture separators.

Turbine problems have been one of the three leading causes of outages of fossil-fired and nuclear power plants. The main materials causes of these outages have involved mainly thermal fatigue cracking of rotors and discs; condensate-related corrosion or stress corrosion cracking of the last stages of the turbine; and solid particle erosion of the first stage guide vanes.

Attempts to correlate the susceptibility to SCC with alloy microstructural differences (segregation/temper embrittlement) in rotors and discs resulting from the initial metallurgical processing routes, or to the operating history of the turbine have not provided much guidance. SCC occurs only in wet steam at crevices or locations where access to the steam is limited, and depends on the contaminants present in the steam. Steam from fossil-fired units invariably picks up impurities from sources such as condensate pump leaks, de-mineralizer/condensate polisher leaks; de-mineralizer breakdown; and from the water treatment chemicals used. Such impurities will deposit from the steam whenever their solubility is exceeded due to changes in steam temperature and pressure. The contaminants most implicated in SCC are usually chlorides, sulfates, hydroxides, and phosphates of sodium and iron.

Since SCWRs are intended to operate essentially continuously near maximum load at temperatures significantly higher than BWRs, it is expected that their potential for solid particle erosion will be similar to that for the present fleet of fossil-fired supercritical steam power plants. The potential for solid particle erosion damage depends on the physical dimensions of the flakes of oxide and the frequency of exfoliation events that, in turn, varies significantly among the alloy types that are used for the upstream piping. Exfoliation is triggered when the stresses in the growing oxide scales exceed some critical value; these stresses result from the thickness of the scale (accommodation between the volume of oxide formed and the volume of alloy consumed), as well as from the mismatch in the coefficients of thermal expansion of the scale and the underlying alloy during cooling from operating temperature. Relationships have been developed for time, oxide-scale thickness, and tendency for scale exfoliation for some of the candidate alloys used in fossil plants, and these can provide guidance on the time at temperature at which exfoliation problems might be expected.

The materials considerations for the SCWR should be based primarily on fossil plant practice, with two caveats:

- i. The maximum alloy temperature required in the SCWR is not higher than the maximum alloy temperature allowed in fossil service
- ii. The threat of stress corrosion cracking (SCC) from oxidizing or other species resulting from radiolysis of the water is not greater than that from the water conditions prevailing in the supercritical fluid in fossil plants.

4 BUDGETS

Table VIII shows the SCWR required budget. The budget figures represent an upper boundary to the actual costs, as credit is not taken for (1) cost-sharing with other Generation IV reactor development programs, and (2) the systematic use of universities to perform key R&D. It is likely that the actual U.S. required budget to perform the R&D described in this report will be lower than the figures indicated in Table VIII.

DRAFT

Table VIII. Required SCWR Budget (\$K).

DRAFT

5 MILESTONES

5.1 System Design and Evaluation

FY 2004

- Identify and evaluate suitable safety systems for the total loss of feedwater transient.
- Complete pre-conceptual design of the steam and pressure relief system, and residual heat removal system.

FY 2005

- Complete design of the test-section for the Erlangen facility.
- Complete pre-conceptual design of ECCS and containment.

FY 2006

- Complete preliminary stability analysis including multi-channel start-up and transient overpower effects.
- Complete construction and shipment of the test-section for the Erlangen facility.
- Complete pre-conceptual design of the core and vessel internals.
- Complete design and construction of the choked-flow facility.

FY 2007

- Complete stability experiments.
- Complete choked-flow experiments.

FY 2008

- Complete heat transfer experiments with surrogate fluids.
- Complete conceptual design of control and start-up systems.

FY 2009 and 2010

- Complete SCW heat transfer experiments in Erlangen.
- Complete stability analysis.

FY 2011 and beyond

- Complete development of heat transfer predictive tools for prototypical SCWR conditions.
- Complete conceptual design of the SCWR including core, reactor coolant systems, safety systems, and containment.

5.2 Materials

The R&D associated with these milestones is described in Section..., and Appendix... (see Corwin's sections)

FY 2004

- Perform a pre-conceptual design of the coolant chemistry control strategy.

FY 2006

- Evaluate steel making and fabrication capabilities for RPV design with current LWR RPV steels.

DRAFT

- Compile available unirradiated mechanical properties data for candidate RPV internals materials.
- Identify most likely construction codes for major components of the steam line and feedwater line piping systems.

FY 2007

- Complete determination of unirradiated mechanical properties data for candidate RPV internals materials.
- Complete compilation of available information on solubility of candidate materials in supercritical steam.

FY 2008

- Complete corrosion and SCC screening tests of unirradiated materials in supercritical water.
- Perform FAC and corrosion fatigue testing for pump materials in supercritical water at simulated chemistry.
- Complete measurements of solubility of candidate materials in supercritical steam.
- Complete evaluation of factors affecting condensation and solubility of corrosive species in steam turbines.

FY 2009 and 2010

- Complete corrosion and SCC testing of primary candidate materials for core support components in supercritical water at simulated chemistry.
- Perform FAC and corrosion testing of secondary candidate materials in supercritical water at simulated chemistry.
- Complete collection and evaluation of data on particle erosion in supercritical steam from fossil experience.

FY 2011 and beyond

- Complete demonstration of fabrication capability for RPV thickness
- Complete irradiation of replaceable fuel assemble candidate materials with neutrons and protons
- Complete post-irradiation mechanical properties testing, microstructural characterization of replaceable fuel assemble candidate materials.
- Complete post-irradiation corrosion and IASCC testing in supercritical water testing of replaceable fuel assemble candidate materials.
- Complete irradiation of candidate materials in supercritical pumped flow loop, post-irradiation mechanical properties testing, microstructural characterization, and corrosion and IASCC testing in supercritical water.
- Complete evaluation of potential for creep-fatigue, thermal fatigue, and dissimilar metal weld cracking of steam line piping valves.
- Complete fatigue, thermal fatigue, and fatigue crack growth testing in simulated supercritical water at simulated chemistry.
- Complete development materials data needed to modify ASME and related construction codes for extended life and new materials.
- Complete development of continuum damage models for steam line piping materials.
- Complete evaluation of potential for dissimilar metal weld cracking in steam line piping.
- Complete testing to predict oxide scale growth, frequency and mode of scale spallation.